NON-PUBLIC?: N
ACCESSION #: 9212150200
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 06

DOCKET NUMBER: 05000410

TITLE: Reactor Scram Caused by Low Vessel Level due to a
Component Failure Followed by a Second Reactor Scram
Signal due to a Level Transient
EVENT DATE: 11/04/92 LER #: 92-022-00 REPORT DATE: 12/04/92

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Operations NMP2

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JC COMPONENT: RLY MANUFACTURER: G080
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On November 4, 1992, at 1011 hours, Nine Mile Point Unit 2 (NMP2)
experienced actuation of an Engineered Safety Feature (ESF). One-quarter
of the Control Rods scrammed when a Reactor Protection System (RPS) K14K
relay failed, after a "half scram" was inserted in the other RPS channel
in an attempt to de-energize this failing relay. The resultant down
power transient caused reactor water level to drop below the low level
scram setpoint causing the full scram. The mode switch was in the "RUN"
position (Operational Condition 1), and the plant was operating at 100
percent rated thermal power. A second reactor scram signal occurred when
a Safety/Relief Valve was cycled shut causing reactor water level to drop
again below the low level scram setpoint.

The root cause of the initial scram signal was failure of an RPS relay.

The cause of the second scram signal was a level transient during SRV cycling.

The Operators immediately took all the actions to complete the plant shutdown. Additional corrective actions included replacing the faulty relay and placing identification placards to identify the K14 relays.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On November 4, 1992, at 1011 hours, Nine Mile Point Unit 2 (NMP2) experienced actuation of an Engineered Safety Feature (ESF). One-quarter of the Control Rods scrammed when a Reactor Protection System (RPS) K14K relay failed after a "half scram" was inserted in the other RPS channel in an attempt to de-energize the smoking relay. The resultant down power transient caused reactor water level to drop below the low level scram setpoint, causing the full scram. The mode switch was in the "RUN" position (Operational Condition 1), and the plant was operating at 100 percent rated thermal power.

Prior to the event, at 1006 hours, Control Room Operators noticed smoke coming from Control Room RPS panel 2CEC*PNL611. This panel houses RPS channel "B" trip logic and instrumentation. The Station Shift Supervisor (SSS), notified of this condition, ordered all personnel to back out of all surveillances ongoing at the time. Arriving at the RPS panel the SSS verified the smoking relay to be a K14 relay but couldn't readily locate the component designation number. However, because the relay was located in panel 2CEC*PNL611 and because of the color coding of the wires going to the relay, it was believed to be associated with RPS channel "B".

At 1008 hours, in an effort to de-energize the smoking relay, the SSS ordered a half scram inserted on the RPS channel "B". This action was intended to de-energize the relay and minimize the probability of a halon Fire Suppression System initiation in the Control Room. After the half scram was inserted, the SSS felt the relay housing and could not determine if this action had actually de-energized the relay. At this point the designator for the relay was found and it was determined to be C72-K14K.

At 1010 hours the SSS, with assistance from instrument and Control Technicians, discovered that relay C72-K14K was powered from RPS channel "A". Realizing the half scram had been inserted on the wrong RPS channel, the SSS' next action was to order the manual half-scram reset for RPS "B" so that a half-scram could be inserted on RPS "A". Before he had time to give the order the CSO announced that there was a One Quarter Core Scram. This occurred at 1011 hours when the fuse for C72-K14K relay opened. Approximately 10 seconds later and prior to a manual scram insertion, a full scram occurred due to the reactor vessel level shrinking below the low level scram setpoint. The shrink was caused by the down power transient (from 100 to 14 percent thermal power) from the quarter scram.

Subsequent to the scram, a large cooldown rate developed. After efforts to reduce the steam loads did not result in turning the downward pressure trend, the inboard Main Steam Isolation Valves (MSIV) were shut to prevent exceeding Technical Specification cooldown rate limits. While alternate means of pressure control were being brought into service, Steam Condensing mode of the Residual Heat Removal System (RHS) and the Reactor Core Isolation Cooling System (ICS)!, the Safety/Relief Valves (SRV) were being used to control reactor pressure. The ICS was started to provide a steam load and lined up to pump water from the condensate storage tanks back to the condensate storage tanks. Recent changes

I. DESCRIPTION OF EVENT (cont.)

to Operating Procedure N2-OP-31, "Residual Heat Removal System," to prevent isolating the steam supply to ICS turbine on a false excess flow trip signal, extended the time required to warm and start RHS in the steam condensing mode. During the initial phase of the transient, steam demand by the ICS turbine was not enough to maintain pressure within the designated pressure band, and the SRVs had to be cycled to control reactor pressure.

Initially, reactor water level was high in the designated band. When an SRV was opened to reduce pressure, reactor water level would swell above the high level setpoint, which caused the ICS turbine to trip. The ICS turbine tripped and was restarted three times. ICS turbine operation was important at this time because its steam load reduced the number of SRV cycles. In order to reduce reactor water level, the scram was reset at 1115 hours. Resetting the scram resulted in reduced Control Rod Drive System (CRD) flow to the reactor vessel, which in turn allowed the Operator to control water level lower in the designated band.

At 1205 hours after the sixth cycling on the SRVs the vessel level dropped below the low reactor vessel level trip setpoint. This resulted in a second automatic reactor scram signal.

II. CAUSE OF EVENT

A root cause investigation has been performed utilizing Nuclear interfacing Procedure NIP-ECA-02, "Root Cause Evaluation".

The root cause for this event was determined to be an equipment failure. RPS relay C72-K14K began to fail which heated the coil and gave off smoke. Ultimately the relay caused the power supply fuse for C72-K14K to blow. A contributing factor to this event was inadequate human factors consideration for the placement of the relay identification label. This led the SSS to insert the "half scram" on the incorrect channel of RPS. The "half scram" along with de-energizing the C72-K14K relay caused one-quarter of the Control Rods to scram, which led to the automatic scram on low reactor water level.

The cause for the second scram signal was a reactor water level transient. The transient was normal shrink and swell due to cycling SRVs for pressure control. In order to maintain the ICS turbine in operation, Operators intentionally lowered water level within the standard level band on orders from the Assistant Station Shift Supervisor (ASSS) who was in charge of implementing the Emergency Operating Procedures. As water level dropped, the shrink due to shutting the SRV eventually caused the low water level scram signal. Although level was continuously monitored and was not intentionally lowered below the scram setpoint, the SRV cycling resulted in level swings that still caused ICS turbine trips. As the Operators continued to adjust level within the band to prevent the ICS trips, the sixth SRV cycle did drop level below the scram setpoint. Reactor water level was then increased within the level band to prevent any further level scram signals.

III. ANALYSIS OF EVENT

These two events are considered reportable in accordance with 10CFR 50.73 (a)(2)(iv), "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

The one-quarter scram event was bounded by the Control Rod Maloperation analysis in the NMP2 Updated Safety Analysis Report (USAR), Section 15.4.3. A further analysis of the affect on the fuel was performed
y
the fuel vendor. Since the control rods that were inserted were spread throughout the core, the down power transient occurred uniformly across the core. The event was quickly (within 10 seconds) terminated by a full

scram. General Electric has concluded that no fuel integrity concerns exist for this event.

The two reactor scram signals were conservative responses to lowering reactor vessel water level anticipating a loss of coolant inventory accident. In neither case was there a loss of inventory accident, and the reactor scrams posed no safety consequences to the general public or plant personnel.

Immediately following the event, there was some concern that the magnitude of the level swings due to cycling SRVs was excessive. Design Engineering determined that the magnitude was normal and due to reduced reactor pressure, which caused the SRVs to be open longer to reduce pressure. This allowed a greater inventory loss (less heat stored in each Ibm of steam) as well as more time for level to shrink and swell. The magnitude was determined not to be due to other causes, including gases coming out of solution in the level instrument sensing lines.

The duration of the first scram signal was 1 hour and 4 minutes. The duration of the second scram signal was 41 minutes.

IV. CORRECTIVE ACTIONS

The Control Room Operators' immediate actions included entering the Emergency Operating Procedure N2-EOP-RPV, "RPV Control" to restore reactor water level, and completing the plant shutdown per Operating Procedure N2-OP-101 C, "Plant Shutdown".

Additional corrective actions include:

1. Relay C72-K14K was replaced along with the blown power supply fuse. The failed relay was sent offsite to an independent laboratory for failure analysis. An inspection of all K14 relays for overheated coils was performed.

IV. CORRECTIVE ACTIONS (cont.)

2. Placards have been installed to clearly identify each of the K14 relays as well as its power supply.

3. Operating Procedure N2-OP-101 C, "Plant Shutdown" will be revised to clarify the use of the mechanical vacuum pumps to establish condenser vacuum, allowing use of the Main Steam line drains to aid controlling reactor pressure with the MSIVs shut.

4. The Training Department will remodel the simulator using actual plant data from this event. Then simulator training on the affects of reduced pressure on shrink and swell will be performed. In the interim, a Lessons Learned Transmittal will be issued to alert Operators to the magnitude of reactor water level swings at reduced pressures.

V. ADDITIONAL INFORMATION

A. Failed component identification:

Component description - Contactor
Mark number - K14K-2RPSB01
Manufacturer - General Electric
Part number - 145C3209P005
Symbol number - 93-96-296
Niagara Mohawk drawing - 793E766
Niagara Mohawk spec - P800A

B. Previous similar events:

NMP2 has experienced one-quarter core scram events twice in the past (LERs 86-010 and 86-014), however, both of these events occurred while the reactor was shutdown and were not due to a failed K14 relay. The corrective actions from these two previous events would not have prevented this event from occurring. GE SIL No. 508 addressed the potential for premature failure of model CR105 relay coils. NMP2 reviewed the SIL and determined that our relays did not fall under the concerns of the SIL (lower ambient temperatures and shorter service life). Currently, Niagara Mohawk management is re-evaluating the preventive maintenance of GE model CR105 relay coils. As an interim action, Maintenance is monitoring these relays for increased temperature by visual inspection and thermography.

V. ADDITIONAL INFORMATION (cont.)

C. Identification of components referred to in this LER:

IEEE 803 IEEE 805
COMPONENT EIIS FUNCTION SYSTEM ID

Reactor Protection System N/A JC
Residual Heat Removal System N/A BO
Reactor Core Isolation Cooling System N/A BN
Control Rod Drive System N/A AA
Contactor RLY JC
Main Steam Isolation Valves ISV SB
Safety/Relief Valves RV SB
Condensate Storage Tank TK KA
Turbine TUR BN
Panel PL JC
Fuse FU JC
Reactor Vessel Level Indicator LI AC
Drain Valves V SB

ATTACHMENT 1 TO 9212150200 PAGE 1 OF 1

NIAGARA
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Neil S. "Buzz" Carns
Vice President December 4, 1992
Nuclear Generation NMP87296

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 92-22

Gentlemen:

In accordance with 10CFR 50.73, we hereby submit the following Licensee Event Report:

LER 92-22 Is being submitted in accordance with 10CFR 50.73 (a)(2)(iv), "any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR 50.72 (b)(2)(ii) report was made at 1125 hours on November 4, 1992. A second 10CFR 50.72 (b)(2)(ii) report was made at 1228 hours on November 4, 1992.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Mr. N. S. Carns
Vice President - Nuclear Generation

NSC/RLM/lmc
ATTACHMENT

pc: Mr. Thomas T. Martin, Regional Administrator Region I
Mr. Wayne L. Schmidt, Senior Resident Inspector

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